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Document Control Desk
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

ATTENTION: MR. D. M. CRUTCHFIELD

SUBJECT: CLOSURE OF AP600 TESTING PROGRAM ISSUES

Dear Mr. Crutchfield:

In your letter to Westinghouse dated November 4, 1993, you noted that Westinghouse and the staff have satisfactorily resolved a number of test program issues related to the details of facility design, test program plans, and program reporting requirements. The letter noted, however, that several test program details did not appear to be moving towards resolution in a manner which would support the staff's AP600 review schedule.

Westinghouse has been working closely with NRC staff to resolve remaining test program issues. As a result of these interactions, a number of these issues have since been resolved. A path to resolution has been identified for the few remaining issues and are being closely pursued to closure. During an April 7, 1994 meeting, Westinghouse and NRC staff discussed each of the test program issues. The attachment to this letter provides the current status of each of the issues identified in the November 4, 1993, letter and incorporates Westinghouse's understanding of the mutual conclusions reached at the April 7, 1994 meeting.

Please contact Brian A. McIntyre on (412) 374-4334 if you have any questions concerning this transmittal.

Br A McIntyre for

N. J. Liparulo, Manager
Nuclear Safety & Regulatory Activities

/nja

Attachment

cc: T. Kenyon, NRC
R. Hasselberg, NRC
R. Borchardt, NRC

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1. CORE MAKEUP TANK TESTS

1A. CMT TEST MATRIX

Issue Description from 11/4/93 letter
The CMT must perform its function of delivering water to the downcomer a) while reactor coolant circulates from the cold leg to the upper level of the CMT, b) while a mixture of hydrogen and steam flows from the pressurizer to the upper level of the CMT, and c) while pressurized cold water and then nitrogen flow from an accumulator into the same delivery line that connects the CMT to the downcomer. The current test matrix does not address the complex behaviors concomitant with these interactions. The staff has suggested changes in the test matrix that have not been made.

Westinghouse Response and Actions Taken
The NRC has provided Westinghouse with a number of comments and suggestions on the design and operation of the Core Makeup Tank separate effects tests. In response to these comments, Westinghouse has made modifications to the instrumentation and design of the CMT test facility and has revised the CMT test matrix. These changes have been discussed with NRC staff at meetings on December 10, 1993 and March 14, 1994. The changes have been documented in responses to NRC Requests for Additional Information (RAI) and in Revision 2 of WCAP-13345, "AP600 Core Makeup Tank Test Specification," which was transmitted to the NRC via Westinghouse letter NTD-NRC-94-4068, dated February 22, 1994. The issue was discussed with NRC staff on April 7, 1994. During the April 7, 1994 meeting, Westinghouse discussed a change to the AP600 design which eliminates the pressure balance lines between the pressurizer and CMTs. This change was made to simplify the plant and reduce capital costs. As a result of this change, several CMT tests, focusing on the effects of the pressurizer balance line, are no longer relevant. These tests will be removed from the CMT test matrix.

At the April 7, 1994 meeting, NRC staff agreed that the test matrix issues raised in the November 4, 1994 letter had been adequately addressed. Westinghouse agreed to provide the revised CMT test matrix once changes have been made to reflect the deletion of the pressurizer balance line.

Current Status
Based on the actions taken and feedback from NRC staff at the April 7, 1994 meeting, Westinghouse believes that the issues related to the CMT test matrix have been adequately addressed and that the issues raised in the November 4, 1993 letter are closed. The action remains to provide the revised CMT test matrix once changes have been made to reflect the deletion of the pressurizer balance line.

The Core Makeup Tank tests are currently in-progress. Information gained from completed CMT tests, as well as information from tests performed at two AP600 integral systems test facilities (SPES-2 and OSU) will be assessed to ensure that the CMT tests continue to provide the information necessary for code validation.

1B. CMT SCALING ANALYSIS

Issue Description from 11/4/93 letter
None has been provided in response to the staff's request.

Westinghouse Response and Actions Taken
The requested scaling analysis for the Core Makeup Tank test facility, WCAP-13963, "Scaling Logic for the Core Makeup Tank Test," was provided to the NRC via Westinghouse letter NTD-NRC-94-4068, dated February 22, 1994. The scaling analysis was discussed during a March 14, 1994 meeting with NRC staff.

Current Status
Westinghouse believes that this issue has been adequately addressed with the issuance of WCAP-13963. At the April 7, 1994 meeting, NRC staff agreed that the general issue raised in the November 4, 1993 letter had been adequately addressed. Westinghouse has the action to respond to comments received from the NRC on the CMT scaling analysis. Westinghouse is currently preparing responses to these comments.

2. SPES-2 TESTS

2A. TEST MATRIX

Issue Description from 11/4/93 letter
The following accidents have been identified by the staff for simulation in the SPES facility because they represent significant plant behavioral questions that must be examined and understood in order to certify the AP600: 1) a steam generator multi-tube rupture sufficient to result in automatic depressurization system (ADS) actuation or which leads the operator to initiate ADS; 2) a station blackout, resulting in a cooldown to "safe-shutdown" conditions followed by ADS actuation after a simulated 24 hours' 3) anticipated transient without scram (ATWS) [SPES is the only AP600 scaled facility capable of scaling full power]; 4) the effects of hydrogen collected above the pressurizer on the progress of a number of accident scenarios.

2A.1 SPES-2 SGTR leading to ADS actuation

Westinghouse Response and Actions Taken
The SPES-2 test matrix includes a beyond design basis test to simulate three failed steam generator tubes. The objective of this test is to characterize the facility response to the increased number of failed tubes. If the event results in sufficient drainage from the CMT, the Automatic Depressurization System (ADS) will actuate. This test was added to the SPES-2 test matrix to address NRC questions on the response of the AP600 for a steam generator tube rupture event in which ADS is actuated.

Subsequent plant calculations with NOTRUMP indicate that for five (5) failed tubes, the CMT will not drain to the level setpoint which would result in ADS actuation. A change to the CMT level setpoint for ADS actuation from 75% to 67% provides additional margin to ADS actuation.

Since it appears that a multiple tube rupture test with a reasonable number of failed tubes will not activate ADS, it is proposed to perform a single tube rupture test. After CMT level recovery has started to occur during the test, ADS would be manually actuated from the control panel.

Current Status
Westinghouse will perform pre-test predictions for the single tube rupture test with ADS activation. Provided that there are no facility problems or limitations, the single tube rupture test with manual ADS activation will replace the multiple tube rupture test in the SPES-2 test matrix. This approach was

discussed with NRC staff on April 7, 1994. At this meeting, NRC staff agreed that the proposed test, modeling a single SGTR with manual ADS actuation, is an acceptable approach. An action was taken by NRC staff to provide a recommendation for the timing for manual ADS actuation in the test.

2A.2 SPES-2 Station blackout conditions followed by ADS

Westinghouse Response and Actions Taken

This issue was discussed at the December 10, 1993 meeting between Westinghouse and NRC. At this meeting, Westinghouse presented information on the calculated response of the AP600 to a Station Blackout event; identifying the plant conditions at the time of automatic ADS initiation. The discussion also identified the SPES-2 tests which will provide useful information relevant to the conditions and phenomena introduced by a Station Blackout event. The addition of a Station Blackout simulation to the SPES-2 test matrix will provide limited useful code validation information above that which will be provided by the planned tests. The following discussion elaborates on the reasons why the Station Blackout simulation is not warranted.

The AP600 automatic depressurization system (ADS) will actuate automatically twenty-four hours after a station blackout event. At that time the plant conditions will be benign relative to the full power condition, so the ADS behavior during such an event is bounded in severity by the inadvertent ADS scenario analyzed in the SSAR from a full power condition.

During the twenty-four hours of station blackout the core makeup tanks will effectively reach thermal equilibrium with the reactor coolant due to cold leg balance line recirculation. After twenty-four hours of shutdown operation (with the PRHR in operation after receipt of a steam generator secondary side low level signal) the reactor coolant system temperature and pressure will be approximately 350 °F and 300 psia respectively. Upon actuation of ADS, the liquid in the core makeup tanks will flash as the system depressurizes. This flashing will serve to accelerate the CMT draindown, but it does not represent a significant challenge to the AP600 passive safeguards system capability. In the one-inch break LOCA case reported in the SSAR, flashing of hot liquid occurs in the CMTs during much of the tank draining transient. Results obtained for the one-inch break are favorable relative to larger postulated LOCA cases.

The same thermal-hydraulic phenomena observed in the inadvertent ADS actuation case in the SSAR are expected to occur in the station blackout event, except that the reduced primary side temperature and pressure of the latter event would result in a more gradual depressurization than in the full pressure/temperature SSAR case. The flashing behavior of the core makeup tanks will be examined over the pressure range of the station blackout event not only in the CMT component separate effects tests but also in the Oregon State University (OSU) integral systems test facility. The initial OSU facility test system pressure will be 385 psia. Performance of the PRHR when the IRWST is saturated has already been examined in the PRHR separate effects testing.

Achieving IRWST injection following ADS actuation is easier for the twenty-four hour station blackout case than it is for the inadvertent ADS actuation from full power presented in the SSAR. Not only is the RCS temperature much lower for the station blackout, but the decay heat present after 24 hours is less than 40% of the core decay heat at the time of IRWST injection initiation in the SSAR inadvertent ADS actuation case. The reduced core steam generation at 24 hours is a major effect that makes IRWST injection much easier to achieve. Injection is not significantly impeded by the fact that the IRWST level has decreased by at most three feet over the 24 hour period.

The recent ADS changes described in "AP600 Design Change Description Report" (Westinghouse letter NTD-NRC-4064, dated 2/15/94), include a 50% larger fourth stage ADS vent area when a single failure is assumed. The increased ADS fourth stage effective vent area, together with the lower decay heat and lower stored energy in the AP600 system at 24 hours, means that achievement of IRWST injection in automatic station blackout actuation is bounded by the inadvertent ADS actuation SSAR case.

Current Status

The station blackout 24-hour actuation of ADS represents a simple depressurization of the RCS from relatively low pressure (< 400 psia). The AP600 test program includes small break LOCA tests which will exhibit CMT heatup over an extended period of cold leg recirculation. Any significant system effects of heated CMT fluid flashing during depressurization will be observed in these tests. The SPES-2 test matrix includes pre-operational tests which actuate ADS from both full and low pressure, so the impact of initial RCS conditions on ADS performance will be observable. Steam generation upon ADS actuation into a saturated, containment pressure IRWST is not an issue because the steam will bubble out the top of the tank. In addition, the pressure range of the OSU matrix tests covers the expected pressure range for ADS depressurization. Overall, adequate information on ADS actuation from different conditions will be available, and additional useful data would not be obtained from a test simulating the 24 hour station blackout ADS actuation.

The Westinghouse position, as outlined above, is that the relevant phenomena of interest are already addressed by the planned tests and there is no need for an additional test to address the requested station blackout scenario. This position was discussed with NRC staff on April 7, 1994. At this meeting NRC staff stated that they were awaiting the results of calculations being performed by INE'.. NRC staff took the action to review the Westinghouse submittals and the INEL calculations. This issue remains open until the NRC review is completed.

2A.3 SPES-2 ATWS

Westinghouse Response and Actions Taken
This issue was discussed at the December 10, 1993 meeting between Westinghouse and NRC. At this meeting Westinghouse presented information on the calculated response of the AP600 to an ATWS event. While the SPES-2 facility is the only test facility that starts the transients from scaled full power, in an ATWS transient the core power is a dependent, not independent, parameter. The core power depends on the fluid conditions such as the moderator temperature coefficient and the resulting neutronic feedback which will limit the core power. The power conditions are not known prior to the transient such that it would be difficult, if not impossible, to simulate these conditions in the SPES-2 facility. A more direct approach is to perform calculations for the ATWS transient for the AP600 to see if the fluid conditions and thermal-hydraulic phenomena are significantly different from the planned experiments. A main feature of the ATWS transient is the CMT recirculation at medium and high pressures. This effect will be covered in the small break LOCA transients, and the steam generator tube rupture tests. Therefore, there is no need to specifically model an ATWS transient in the SPES-2 facility.

Current Status
Westinghouse believes that the plant conditions and phenomena resulting from an ATWS event will be adequately represented by the planned tests in the SPES-2, OSU and CMT test facilities. An ATWS would be difficult to simulate in the SPES-2 facility and would provide limited useful data for code validation. The limiting AP600 ATWS transient behavior were provided in the response to RAI 440.26. The calculations indicate that the CMT recirculates and is injecting at pressures from 2250 to 1200 psia. The majority of this pressure range will be covered in the CMT separate effects tests as well as the SPES-2 small break LOCA and steam generator tube rupture tests. Since CMT recirculation is single phase fluid, the extrapolation to higher pressures can be made with confidence.

The Westinghouse position, as outlined above, is that the relevant phenomena of interest are already addressed by the planned tests and there is no need for an additional test to address the requested ATWS scenario. This position was discussed with NRC staff on April 7, 1994. At this meeting NRC staff stated that they were awaiting the results of calculations being performed by INEL. NRC staff took the action to review the Westinghouse submittals and the INEL calculations. This issue remains open until the NRC review is completed.

2.A.4 Effects of hydrogen

Westinghouse Response and Actions Taken
This issue was discussed with NRC staff at a meeting on December 10, 1993. At this meeting, NRC staff

clarified the issue by stating that the NRC concern is noncondensible gas collection in the CMT which has the potential to block recirculation. Since this meeting a design change has been made which deletes the pressure balance lines from the pressurizer to the CMTs. This change, discussed with NRC staff on April 7, 1994, removes the hydrogen concerns addressed in the November 4, 1993 letter.

Current Status
The recent design change to remove pressurizer to CMT balance lines removes the hydrogen concern identified in the November 4, 1993 letter. At the April 7, 1994 meeting, NRC staff agreed that this issue is adequately addressed by the design change.

3. OREGON STATE UNIVERSITY (OSU) TESTS

3A. TEST MATRIX

Issue Description from 11/4/93 letter
For the same reasons, the following accidents should be simulated within the OSU matrix: 1) capability of the safety and non-safety systems to deal with higher risk shutdown incidents; 2) the effects of hydrogen collected above the pressurizer and nitrogen in the accumulator on the progress of a number of accident scenarios; 3) to allow for the possibility that SPES shows that steam generator tube rupture (SGTR) or main steamline break (MSLB) accidents can lead to ADS actuation, the capability of performing such tests should be provided.

3A.1 OSU simulation of higher risk shutdown incidents

Westinghouse Response and Actions Taken
The design of the AP600 systems addresses lessons learned from incidents that have occurred during shutdown that had the potential to challenge core cooling. In addition, the AP600 PRA evaluates the design at shutdown conditions. Both of these factors were considered in the design of the passive safety-related and the nonsafety-related systems.

The basic design approach for the AP600 is to provide safety-related emergency core cooling during all modes of plant operation including shutdown. The safety-related systems relied upon are covered by the technical specifications for the required modes of operation. In addition, the nonsafety-related systems that normally cool the reactor during shutdowns are designed to avoid issues identified by incidents in operating plants.

This issue was discussed with the NRC staff on December 10, 1993. At this meeting the staff indicated that they were just beginning to review the AP600 design and did not fully understand the different ways the AP600 responds to shutdown events. It was also indicated that the reviewers would become available soon and would be giving their attention to AP600. Another meeting was held with the staff on March 10, 1994 to specifically discuss shutdown events. During this meeting Westinghouse discussed in detail the different modes of shutdown operation and which of the passive safety-related systems is relied upon. The nonsafety-related systems were also discussed.

As a part of the meeting Westinghouse presented levels of defense flow charts for six different shutdown events, three of which were from mid-loop conditions. These charts clarified the relative roles of the safety and nonsafety-related systems. It also demonstrated that there was always a safety-related system available. The flow rates / pressure drops in the IRWST gravity injection / ADS valves that provide safety-related cooling during mid-loop operation was compared to that seen during a spurious ADS. This comparison shows that the spurious ADS has greater flow rates with similar DPs. In addition, there is more non-condensible gas present during IRWST initiation because of the accumulators. This supports the use of the current OSU test matrix to provide information to validate the AP600 computer codes. It should also be noted that the OSU test matrix has a case where the RNS is actuated following spurious

ADS from full power conditions.

The March 10, 1994 meeting also included discussion on the design features of the nonsafety-related systems that improve their shutdown operation. Particular attention was given to the improvements made to the normal RHR system. Results from the RTNSS evaluation (WCAP-15856) show that these nonsafety-related systems were not important to meeting the NRC safety goals. The RTNSS results also show the RNS and its supporting systems to be important to the mid-loop initiating event evaluation. As a result Westinghouse proposed in the RTNSS evaluation that there be availability controls that require both trains of RNS to be availab. during mid-loop.

Current Status
As discussed above, Westinghouse believes that the at-power OSU matrix tests will provide adequate information to qualify the computer codes for shutdown conditions. This issue was discussed with NRC staff on April 7, 1994. At this meeting NRC staff agreed that the Westinghouse approach adequately addresses this issue. Should results from testing identify unusual or unexpected behaviors that could impact operations during shutdown conditions then Westinghouse will evaluate these results to determine whether additional testing is warranted.

3.A.2 Effects of hydrogen and nitrogen

Westinghouse Response and Actions Taken
This issue was discussed with NRC staff at a meeting on December 10, 1993. At this meeting, NRC staff clarified the issue by stating that the NRC concern is noncondensible gas collection in the CMT which has the potential to block recirculation. Since this meeting a design change has been made which deletes the pressure balance lines from the pressurizer to the CMTs. This change, discussed with NRC staff on April 7, 1994, removes the hydrogen concerns addressed in the November 4, 1993 letter. The effects of Nitrogen on CMT operation are being addressed parametrically in the CMT separate effects tests. Additional information of the effects of Nitrogen will be obtained from integral system testing at SPES-2 and OSU Both of these test facilities include the simulation of Nitrogen gas discharge from the accumulators.

Current Status
The recent design change to remove the pressurizer to CMT balance lines removes the hydrogen concern identified in the November 4, 1993 letter. The effects of Nitrogen are also being addressed in CMT separate effects tests and are simulated in both the SPES-2 and OSU integral systems tests. These tests will provide necessary information to address questions on the effects of Nitrogen gas discharge from the accumulators. At the April 7, 1994 meeting, NRC staff agreed that this issue is adequately addressed by the design change and the testing planned at the CMT, SPES-2 and OSU test facilities.

3.A.3 Capability of performing SGTR and MSLB accidents

Westinghouse Response and Actions Taken
This issue was discussed with NRC staff at a meeting on December 10, 1993. At this meeting Westinghouse stated that the full capability for modeling SGTR and MSLB events was not currently provided at the OSU test facility. However, Westinghouse stated that this capability has not been precluded at OSU.

Current Status
While full capability to model SGTR and MSLB events is not currently provided at OSU, this capability has not been precluded. Should results from testing at the SPES-2 test facility identify unusual or unexpected behaviors for these events, Westinghouse will evaluate these results to determine whether additional testing is warranted and whether this testing should be performed at the OSU test facility. This position was discussed with NRC staff on April 7, 1994. At this meeting, NRC staff stated that this position adequately addresses the issue raised in the November 4, 1993 letter.

4. PASSIVE RESIDUAL HEAT REMOVAL (PRHR) HEAT EXCHANGER

4A. PERFORMANCE DATA

Issue Description from 11/4/93 letter
The C-shape of the heat exchanger tubes and their arrangement in the in-containment refueling water storage tank (IRWST) differ markedly from the 3-vertical tube test used to develop the description of the PRHR performance. Justification is needed to support Westinghouse's claim that no more testing is required, especially considering pool-side thermo-hydraulics.

Westinghouse Response and Actions Taken
This issue was discussed at the December 10, 1993 meeting between Westinghouse and the NRC and additional information on the PRHR tests and their applicability has been provided in response to NRC Requests for Additional Information (RAI 952.14 and RAI 440.13).

The range of conditions which were examined in the PRHR test program were given in the Response to RAI 440.13 and compared to the plant calculations for the C-tube PRHR. As the tables in this response indicated, the range of test conditions covers the calculated plant conditions for the PRHR.

The top and bottom portions of the heat exchangers are horizontal. Comparisons of boiling heat flux between horizontal and vertical surfaces are shown in Figure 6 of the response to RAI 952.14. As shown in the figure, for the same wall superheat, the heat flux on the horizontal surface is higher, therefore, the vertical heat transfer data from the PRHR tests is a conservative estimate of the horizontal tube heat transfer.

The question on possible dryout on the secondary side of the PRHR has been examined. In reviewing test data, it was observed that the high surface heat fluxes can only occur when the primary flow in the tube is high, that is, under conditions that simulate the operation of the main reactor coolant pumps. The highest heat fluxes occur at the top of the heat exchanger where primary fluid temperature is the highest. The calculated surface heat fluxes for these conditions are between 1×10^5 and 2.5×10^5 btu/hr-ft^2 depending on primary flow and temperature. Dryout was never observed in these cases, and the heat fluxes are below the Zuber pool boiling limit and even further below a flow boiling CHF limit.

When examining the cases which simulate the natural circulation flow on the primary side of the heat exchanger, the calculated peak heat fluxes are significantly lower, in the range of 2.5×10^4 to 7.0×10^4 btu/hr-ft^2 depending on the primary flow rate and fluid temperature. Therefore, even when accounting for multi-tube effects, which have been shown to decrease CHF in a bundle relative to a tube, the natural circulation conditions result in such low peak fluxes that bundle dryout will not occur under these conditions.

Important factors to PRHR performance are the following:

1) The peak heat fluxes occur under conditions when the reactor coolant pumps are operating.
2) PRHR performance is most important after the RCPs are tripped. The design basis accident analyses that credit PRHR performance are events where the CMTs are actuated and the RCPs are tripped. Therefore the PRHR heat fluxes are lower as described above.
3) Neither the SSAR safety analysis codes nor the PRHR test data predict the PRHR will experience film boiling or dryout, even assuming high flow rates due to the RCPs operating.
4) The PRHR tube thickness was specifically selected to reduce the probability of CHF in the PRHR (the PRHR tube thickness is 0.065 inches compared with SG tubes that have a thickness of 0.049 inches.)

If dryout did occur in the PRHR under pumped primary flow conditions; the heat load would move down the tubes and nucleate boiling would be established again and the remainder of the heat exchanger would be more efficient since the primary temperature would be higher. As time progresses, the primary fluid temperature decreases due to the heat removed by the PRHR such that at some point, nucleate boiling would be reestablished on all the tube surface area.

At the December 10, 1993 meeting, Westinghouse was requested to investigate the literature to determine the CHF limit for tube bundles and arrays similar to the C-tube geometry of the revised PRHR. The most limiting position in the heat exchanger is the top where the vertical portion transitions to the horizontal tube bundle. The vapor and two-phase flow developed in the vertical portion will flow across the high temperature and high powered horizontal portion of the heat exchanger. The literature effort has started and papers have been found which indicate the CHF limits for individual horizontal tubes with factors that give the bundle effects. The existing PRHR data has also been reviewed to find the range of expected heat fluxes that would occur during the natural circulation portion of the transient when the PRHR is most important. This was described in the response to RAI 952.14. In addition to the literature search, a simple one-dimensional model is being set-up to calculate the flow and quality at the vertical/horizontal intersection of the heat exchanger to use in CHF correlations. These calculations will help estimate the CHF margin that exist for the heat exchanger.

Current Status

The phenomena of interest for the PRHR have been investigated in the PRHR tests and information available in open literature. The effects due to differences in PRHR configuration are limited and additional testing will not add significantly to this body of information. Additional investigation of the open literature is underway to add to the database of information on CHF limits for tube bundles and arrays. The equations have been formulated for the simple one-dimensional calculation of the fluid conditions in the heat exchanger and are in the process of being programmed. These calculations will provide a quantification of the CHF margin that exists during heat exchanger operation.

This issue was discussed with NRC staff on April 7, 1994. At this meeting, NRC staff stated that they were reviewing the responses to RAIs and PRHR results from the ROSA-V test facility which, like the SPES-2 and OSU tests, employs a "C" tube heat exchanger. Westinghouse has an action to provide the results from the stated literature search. This issue remains open until NRC staff have completed their review of the Westinghouse submittals and other pertinent information.

5. 4TH STAGE AUTOMATIC DEPRESSURIZATION SYSTEM (ADS) VALVES

5A. PERFORMANCE TESTING

Issue Description from 11/4/93 letter

Operation of these valves is critical to the initiation of cooling from the IRWST. Westinghouse has claimed that no test data are required for certification. The staff does not agree with this position. Unless a very conservative analysis can be supplied to support Westinghouse's position, the staff requests the timely submission of a detailed test program that will produce thermo-hydraulic data over a range of containment pressures.

Westinghouse Response and Actions Taken

The Westinghouse position is that design certification testing is not required. This position is based on the fact that the 4th stage is much simpler than the first three stages because it has fewer parallel paths, the downstream piping is completely separate, the discharge paths are straight pipes that discharge into the containment and not under water through a sparger. The following steps will be performed to verify the performance of these valves:

1) A sensitivity study will be performed under design certification, where the capacity of the 4th stage valves are reduced. This study will quantify the margin in the design.
2) The prototype 4th stage valves will be tested during their equipment qualification program. This testing will show that the valves can pass the minimum amount of steam at the limiting RCS pressures. This pressure is anticipated to be in the range of 100 psia.

Westinghouse believes that the plan discussed above is sufficient to demonstrate that the 4th stage ADS valves will perform as required. This issue was discussed with NRC staff on April 7, 1994. At this meeting the NRC staff agreed with the Westinghouse position of performing sensitivity calculations in lieu of 4th stage ADS testing within design certification. The NRC took an action to provide Westinghouse with feedback on what would comprise an acceptable sensitivity study.

There was additional discussion during the April 7, 1994 meeting on the testing to be performed outside of design certification. Comments and actions resulting from these discussions are identified below:

Action:	Westinghouse to provide NRC with a narrative of the valve testing roadmap slide used in the April 7, 1994 meeting. This action applicable to ADS valves and PXS check valves.
NRC Comment:	Prototype tests of the valve assembly need to be performed not only at design basis conditions but also over a range of conditions to allow correlation.
NRC Comment:	Production valve assembly should be tested under some flow if possible. Could possibly reduce testing requirement for startup by doing this.
NRC Comment:	If severe accident conditions are well beyond design conditions then the NRC will need some assurance that the valves will work under severe accident conditions if the function would be relied upon. This comment applicable to ADS valves and PXS check valves.

Current Status:

Sensitivity calculations for the 4th stage ADS valves will be performed as part of design certification. Westinghouse will obtain NRC staff feedback on the sensitivity calculations to be performed. Based on the discussion at the April 7, 1994 meeting, the issue identified in the November 4, 1994 letter is adequately addressed by sensitivity studies. Additional information on the testing to be performed outside of design certification is needed by NRC staff to assess the adequacy of the test plans.

6. CHECK VALVES

6A. PERFORMANCE AND RELIABILITY QUALIFICATION TESTING AND ISI/IST

Issue Description from 11/4/93 letter
Performance of these valves is critical to the CMTs, the accumulators and the IRWST. The staff must also have assurance that those check valves in the IRWST delivery lines will open under a very low delta-P following long periods of operating closed under a large delta-P. Therefore, Westinghouse must provide plans for performance and reliability qualification testing. ISI/IST procedures that are necessary to maintain qualification throughout the design plant life, including any design considerations for inservice testing and provisions for inservice inspections, must be provided.

Westinghouse Response and Actions Taken
This issue was discussed with the NRC staff at a meeting on December 10, 1993. At this meeting, Westinghouse presented its plan which included the following:

1) Simple swing disk check valves are a very good / appropriate choice for the IRWST injection and sump recirculation valves.
2) Simple swing disk check valves used in nuclear service (reactor grade water, stainless steel material, standby service) have been very reliable, i.e., no reported failures to open.
3) Westinghouse performed a test that showed that three of these check valves in series operating over a wide range of flows displayed no adverse modes of operation. This test also showed that these check valves initiated flow with essentially no differential pressure.
4) Westinghouse performed a test at an operating plant that showed that even after more than a year at high differential pressures such check valves operated with acceptable differential pressure. Additional testing is planned.
5) The failure rate used in the AP600 PRA for these check valves is greater than that seen for historical check valves, 1 failure per 115 demands.
6) The inservice testing of these check valves will be performed with prototypical differential pressures and flow rates during every refueling.

Based on comments received from the staff during this meeting, Westinghouse is revising its IST plan. It will now include measurements of the differential pressure required to initiate flow and the flow required to fully open the valves.

This issue was also discussed with NRC staff on April 7, 1994. At this meeting, NRC staff agreed, in principle, with the Westinghouse plans for performance and reliability qualification testing. Westinghouse has the action to provide ISI/IST plans in response to RAI 210.24. Additional comments and actions from the April 7, 1994 meeting are summarized below:

Action:	Westinghouse should consider the potential impacts of in-leakage between pair of check valves on design and operation
Action:	Pressure and temperature transients for check valve design should be clearly documented and test criteria for qualification testing should be submitted.
Action:	NRC needs to know the transient conditions to which the check valves would be subjected to. Staff will review and comment on adequacy.
Action:	Prototype testing should be performed for a range of check valve sizes
Action:	Westinghouse should provide criteria for qualification testing and test criteria for prototype.
NRC Comment:	If severe accident conditions are well beyond design conditions then the NRC will need some assurance that the valves will work under severe accident conditions if the function would be relied upon.

Current Status
This issue was also discussed with NRC staff on April 7, 1994. At this meeting, NRC staff agreed in principle with the Westinghouse plans for performance and reliability qualification testing. Westinghouse has the action to provide ISI/IST plans in response to RAI 210.24.

7. PASSIVE CONTAINMENT COOLING SYSTEM (PCCS)

7A. SCALING/TEST SIMULATION

Issue Description from 11/4/93 letter
Westinghouse has not provided a good justification for the use of the WGOTHIC code to predict AP600 behavior. A detailed scaling analysis which relates the 1/8 scale PCCS tests to the prototype has not been provided. The heat transfer and fluid flow correlations used in the scaling analysis were not developed for those conditions in which they are applied in the WGOTHIC code.

The downcomer region of the outer containment wall plays an important role in establishing natural convection evaporation from the water film, and is an integral component in the containment cooling system. Its omission from the PCCS test program is a serious concern to the staff. Therefore, some tests of the complete system (or possibly a detailed analysis) to demonstrate the expected performance is needed.

Westinghouse Response and Actions Taken
A number of discussions and meetings with NRC staff have been held to better define issues related to scaling of containment phenomena and the use and development of WGOTHIC. A number of items have been addressed by testing. Pending the completion of review of the test data, these items have been adequately addressed to the satisfaction of NRC staff. These include:

- Basis for short term and long term heat sinks and internal volumes simulated in PCCS Large Scale Test (LST);
- Test data accounting for the effect of non-condensibles on condensation rates inside containment;
- Test data accounting for the effects of break release location and orientation.

The following summarizes the status of the key remaining PCCS test and analysis issues and plans for resolution.

Velocity Measurements
The NRC staff is concerned that the measurements of interior velocities in the LST Phase 2 tests are insufficient to support computer code validation. This concern is based in part on preliminary calculations of an LST Phase 2 test by NRC contractors which underpredict heat transfer through the containment shell using the measured velocities from the test. Preliminary results from WGOTHIC calculations have resulted in good agreement with the velocity measurements. During a meeting with NRC staff on April 7, 1994, it was decided that an appropriate way to proceed on this issue would be to schedule a meeting to discuss and compare results from computer code calculations. This meeting is proposed for the end of July, 1994.

The following discussion outlines how the LST Phase 2 velocity measurements will be used in the WGOTHIC validation effort.

The LST Phase 2 tests included instrumentation to measure order of magnitude and direction of interior velocities. The LST database includes a very wide range of steam delivery rates, including very low steam rates that lead to prototypical natural convection patterns, as well as higher rates that bracket the expected AP600 range. The WGOTHIC Large Scale Test (LST) validation objective is to predict containment vessel pressure and internal distributions of temperature and noncondensibles. To achieve this objective the internal velocity field must be predicted well enough to yield accurate heat and mass transfer rates, as well as noncondensable distributions.

The velocity measurements were taken where knowledge of the velocity is most important for predicting the fluid mass, momentum, and energy transport - along the vessel wall. They are located at five different positions 1 to 2 inches from the wall to indicate the local bulk velocity and the flow direction along the dome and vertical wall at several elevations and radial locations.

During testing, functional output was not always available from some of the velocity sensors. The velocity performance for 3 of the 5 meters was degraded by exposure to the steam environment. However, the measurements obtained give an indication of the local bulk velocity along the wall which are useful in the validation, although they are not necessary for the validation.

With proper nodalization to model jets and locations of high concentration gradients, WGOTHIC will adequately predict containment non-condensable distribution, axial wall temperatures, and hence velocity flow fields. The LST extensive instrumentation provides a realm of measurements which will be used for validation, including non-condensible fractions and axial wall temperature distributions. If the vessel pressure, the axial temperature and non-condensable distributions are predicted well, it can be inferred that the flow field is adequately predicted; therefore, the velocity measurements in the LST are sufficient to support WGOTHIC validation.

Scaling
As identified in the November 4, 1993 letter, the NRC staff is looking for a scaling analysis which relates the 1/8 scale PCCS tests to the AP600. Information presented to the staff to date has concentrated primarily on bottom-up scaling of individual phenomena models. Work is underway to provide additional top-down scaling evaluations. This work is scheduled to be provided to NRC staff in draft form by the end of July 1994. It is proposed that a meeting to discuss the scaling report be scheduled soon after the release of the draft report.

AP600 Wetted Fraction
Additional information has been requested to justify AP600 containment water coverage fractions assumed in SSAR Rev 0. Additional information will be provided by Westinghouse on this subject by end of July, 1994. The information will include discussion of full scale water distribution test results and will demonstrate that SSAR Rev 0 coverages bound measured coverage when cold-to-hot effects are considered.

Downcomer Flow
Several concerns have been raised by the NRC staff on the way external air flow is modeled in both the PCCS tests and analyses. In meetings and telecons with the NRC, a further understanding of the basis of the NRC concern on downcomer operation has been reached by Westinghouse and NRC Requests for Additional Information (RAIs) addressing the issues are expected. These issues are being addressed via planned meeting and responses to RAIs.

Current Status
The issues and concerns identified above are being addressed through meetings, information submittals and responses to RAIs. A draft schedule of submittals and meetings is provided in Table 7-1. This schedule, intended only as an outline for the continuing review, will be continually revised as necessary to meet the needs of the NRC review.

Table 7-1
Draft Information Exchange Schedule In Support of AP600 PCCS Review

FORMAT	DATE	TOPICS/AGENDA ITEMS	
Telecons	Weekly	- PCCS Analysis schedule - NRC review status, information needs	- Suggested agenda items forwarded prior to telecons - Format similar to test schedule phone calls
Meeting	Early May, 1994	- DSER/FSER supporting information plans - WGOTHIC development	- PCCS analysis work plans - DSER information exchange schedule/content - Schedule for resolution of technical issues - TOC for June 30, 1994 report - TOC for scaling report - Summary of WGOTHIC development - Relationship to EPRI GOTHIC development programs
Report	June 30, 1994	SSAR Bases for PCCS DBA	
Meeting	July 1994	- PCCS Computer Code Validation - SSAR Containment DBA	- WGOTHIC validation results and status - NRC CONTAIN validation results and status - Review status and future data needs
Report	July 1994	Phenomenlogical Report	Document containment surface wetting basis with respect to requirements and film stability (26620W6106)
Report	July 1994	Draft PCCS scaling report	SASM Component I
Report	August 1994	Phenomenological Report	Convective heat transfer basis with respect to laminar/turbulent flow (26620W6102)
Meeting	August 1994	- PCCS Scaling - PCCS Phenomena - Schedules	- Top down scaling results - Bottom up scaling results - NRC comments on phenomena (bottom up) reports - NRC comments on PCCS scaling report
ACRS mtg.	September 1994	PCCS Scaling	
Report	September 1994	PCCS scaling report	Incorporation of NRC comments
Report	October 1994	Phenomenological Report	Condensation and evaporation mass transfer basis with respect to laminar/turbulent, inclined plane (26620W6103,26620W6104)
Meeting	November 1994	- WGOTHIC model review - Discussion of remaining issues, schedule	Model review prior to start of blind test prediction calculations
Report	December 1994	Phenomenological Report	Internal transient and stratification processes (26620W6201, 26620W6301)